Duke Energy

H. B. Barron
Vice President

Duke Energy Corporation
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-9340
(704) 875-4800 OFFICE
(704) 875-4809 FAX

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U.S. Nuclear Regulatory Commission
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Subject: McGuire Nuclear Station, Unit 1
Docket Nos. 50-369
Licensee Event Report 369/02-01, Revision 0
Problem Investigation Process No.: M-02-1039

Pursuant to 10 CFR 50.73, Sections (a)(1) and (d), attached is Licensee Event Report (LER) 369/02-01, Revision 0, concerning a manual trip of the McGuire Nuclear Station Unit 1 reactor and automatic actuation of the Unit 1 Auxiliary Feedwater (CA) System.

On March 4, 2002, with Unit 1 at 100% power, the 1A Steam Generator (SG) experienced decreasing water level when valves in the main feedwater supply to that SG failed closed upon loss of electrical power to their control circuitry. In response to the decreasing SG level, the Unit 1 reactor was manually tripped (Reactor Protection System actuation). After the trip, the 1A SG experienced a Lo-Lo level condition which automatically started the 1A and 1B CA Pumps (Auxiliary Feedwater System actuation). The loss of electrical power to the control circuitry for the failed closed 1A SG feedwater supply valves was caused by an electrical short in a capacitor located within the cabinet containing the control circuitry for these valves.

This event was initially reported on March 4, 2002 in accordance with the requirements of 10 CFR 50.72 (b)(2)(iv)(B) and 10 CFR 50.72 (b)(3)(iv)(A). This LER is being submitted as per the requirements of 10 CFR 50.73 (a)(2)(iv)(A). This event is considered to be of no significance to the health and safety of the public. There are no regulatory commitments contained in this LER.

HB Barron

H. B. Barron

Attachment

IE22

cc: Mr. L. A. Reyes
U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30323

Mr. R. E. Martin
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

INPO Records Center
700 Galleria Parkway
Atlanta, GA 30339

Mr. S. M. Shaeffer
NRC Resident Inspector
McGuire Nuclear Station

bxc: Dhiaa M. Jamil (MG01VP)
Braxton L. Peele (MG01VP)
Scotty L. Bradshaw (MG01OP)
Guynn H. Savage (EC12X)
Gregg B. Swindlehurst (EC08H)
R. Bruce Travis (MG05SE)
Thomas C. Geer (MG05EE)
Michael S. Kitlan (EC08I)
Ronald F. Cole (EC05P)
H. Duncan Brewer (EC08I)
Richard T. Bond (ON03SR)
Kay L. Crane (MG01RC)
Gary D. Gilbert (CN01RC)
L. E. Nicholson (ON03RC)
Mary J. Brown (PB02L)
Lisa Vaughn (PB05E)
Michael T. Cash (EC050)
(NSRB Support Staff) (EC05N)

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LICENSEE EVENT REPORT (LER)	

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 1	05000 369	1 OF 9

TITLE (4)
Manual reactor trip in response to loss of feedwater valve control power.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	04	2002	2002	– 001 –	00	04	29	2002	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)	50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)	73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)	
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)	
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)	
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
J. W. Bryant, Licensing Engineer	(704) 875-4162

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B6a	JF	CAP	W121	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

Unit Status: At the time of the event, Unit 1 was Mode 1 (Power Operation) at 100 percent power. Unit 2 was in No Mode (fuel offloaded).

Event Description: On March 4, 2002, the 1A Steam Generator experienced decreasing level when its main feedwater supply valves failed closed. In response to the decreasing 1A Steam Generator level, the Unit 1 reactor was manually tripped. After the trip, the 1A Steam Generator experienced a Lo-Lo level condition which automatically started the 1A and 1B Auxiliary Feedwater Pumps. There were no malfunctions of equipment needed to respond to the event. This event is considered to be of no significance to the health and safety of the public.

Event Cause: An electrical short in a capacitor located within the cabinet containing the 1A Steam Generator main feedwater supply valves control circuitry caused a loss of electrical power to the control circuitry for these valves.

Corrective Action: Failed components were replaced. McGuire will evaluate possible design changes to prevent a capacitor failure from causing a reactor trip due to a loss of electrical power to process control cabinet circuitry.

BACKGROUND

The following information is provided to assist readers in understanding the event described in this LER. Energy Industry Identification (EIIS) system and component codes are enclosed within brackets. McGuire system and component identifiers are contained within parentheses.

Main Feedwater (CF)[SJ] Control Regulating Valves:

During high flow conditions when Unit 1 reactor power is greater than or equal to 15%, Unit 1 Main Feedwater Control Regulating Valves (FRV)[FCV] 1CF17AB, 1CF20AB, 1CF23AB and 1CF32AB control feedwater flow to maintain acceptable water level in Steam Generators (SG)[SG] 1D, 1C, 1B, and 1A, respectively. These valves are air-to-open valves which fail closed upon loss of air or loss of electrical power to their control circuits. These FRV's are capable of both automatic and manual operation. The mode of operation is selected at the "Auto/Manual" control station located on Main Control Board MC2 in the Control Room. In both Auto and Manual, these FRV's receive a control signal from the Unit 1 7300 Process Control System (PCS)[JF]. The PCS provides two separate automatic control circuits for each FRV (Normal and Alternate). Only one of the two control circuits will be controlling a FRV at any time. The desired control circuit is selected via the "Norm/Alt" select switch located on Main Control Board MC2 in the Control Room.

Main Feedwater Control Bypass Valves (FCBV)[SHV]:

During low flow conditions when Unit 1 reactor power is less than 15%, Unit 1 FCBV's 1CF-104AB, 1CF-105AB, 1CF-106AB, and 1CF-107AB are used to control CF flow to SG's 1A, 1B, 1C, and 1D, respectively. The FCBV's are air-to-open valves which fail closed upon loss of air or loss of electrical power to their control circuits. These valves are capable of both automatic and manual operation. The mode of operation is selected at the "Auto/Manual" control station located on Main Control Board MC2 in the Control Room. In both Auto and Manual, these FCBV's receive a control signal from the Unit 1 PCS.

Auxiliary Feedwater (CA)[BA] System:

The CA System is designed for operation during plant startup, plant shutdown, and emergency conditions where CF is not available. The Unit 1 CA System contains one turbine driven pump and two motor driven pumps. The 1A and 1B Motor Driven CA (MDCA)[P] Pumps start automatically and provide flow to the Unit 1 SG's upon receipt of a 2 out of 4 channel Lo-Lo level signal in 1 out of 4 Unit 1 SG's.

Reactor Protection (RPS)[JC] System:

The function of the RPS System circuits associated with Lo-Lo SG water level is to preserve the SG heat sink for removal of long-term residual heat. Should a complete loss of CF occur, the reactor is tripped on Lo-Lo SG water level before the SG's are dry. This function is initiated upon receipt of a 2 out of 4 channel Lo-Lo level signal in 1 out of 4 Unit 1 SG's.

7300 Process Control System:

The Unit 1 PCS provides the Normal and Alternate control signals for the Unit 1 FRV's in the Auto mode of operation. In addition, it provides the control signal to these valves when they are in Manual control. All control signals associated with the 1A SG FRV (1CF-32AB) and FCBV (1CF-104AB) are processed through PCS Cabinet 5, Card Frame 3. The remaining Unit 1 FRV's and FCBV's are processed through other PCS cabinets. PCS Cabinet 5, Card Frame 3 also contains circuitry for the Unit 1 Pressurizer (PZR)[PZR] Power Operated Relief Valves (PORV)[RV] discharge temperature loop. PCS Cabinet 5 also provides the control signal for the Unit 1 PZR Pressure Master Controller via Card Frame 6.

Electrical Panelboard KXA is the primary alternating current (AC) power source for Unit 1 PCS Cabinet 5. Panelboard KRB is the backup AC power source for Unit 1 PCS Cabinet 5. Primary and backup power is routed to the card frames within Cabinet 5 through card frame primary and backup power fuses and then through single card fuses to the individual control cards on each card frame.

Note that Card C5-345 on Card Frame 3 contains Capacitor C-105, which provides noise filtering for DC Power supplied to PCS Cabinet 5. The fuse for Card C5-345 is located downstream of Capacitor C-105. Consequently, this card fuse would not provide protection in the event of a failure of Capacitor C-105. Instead that protection would be provided by the upstream Card Frame 3 primary and backup power fuses. This arrangement is a standard Westinghouse design.

EVENT DESCRIPTION

On March 4, 2002, McGuire Nuclear Station Unit 1 was in Mode 1 (Power Operation) at 100% power. FRV's 1CF17AB, 1CF20AB, 1CF23AB and 1CF32AB were in Auto control being modulated by their Unit 1 PCS normal control circuitry to maintain acceptable water level in their respective SG. FCBV's 1CF-104AB, 1CF-105AB, 1CF-106AB, and 1CF-107AB were full open, which is their normal position at 100% power. Preparations were underway for de-energizing Electrical Distribution Center MKB for maintenance, which would result in a loss of Panelboard KRB, the backup AC power source for PCS Cabinet 5. Consequently, prior to de-energizing MKB, Panelboard KXA (primary AC source for PCS Cabinet 5) was verified to be supplying electrical power to Cabinet 5. After this verification was completed, the backup power supply to Cabinet 5 from Panelboard KRB was secured.

The relevant sequence of events is as follows (all times approximate):

08:41:00 PCS Cabinet 5, Card Frame 3 experienced a loss of electrical power.

08:41:11 1A SG FRV (1CF-32AB) went closed from its intermediate modulating position. The FRV's for the remaining three SG's continued to modulate normally to control level in their respective SG.

08:41:24 1A SG FCBV (1CF-104AB) went closed from its full open position. The remaining FCBV's remained in their pre-event fully open position. 1A SG water level began rapidly decreasing from approximately 65%. Upon recognition that 1CF-32AB and 1CF-104AB had closed and that 1A SG water level was decreasing, Operators immediately attempted to restore Auto control of 1CF-32AB by attempting to swap to the alternate PCS control circuitry. These actions were unsuccessful in restoring Auto control of 1CF-32AB. Attempts to take Manual control of 1CF-32AB and 1CF-104AB were also unsuccessful.

Unit 1 PZR Pressure Master Controller went from automatic to manual control without operator action. Unit 1 PZR PORV high discharge temperature alarm was received. This alarm was an indication problem only and was not associated with an actual high temperature condition.

08:41:40 After 1A SG water level had decreased to 35%, Operators appropriately tripped the Unit 1 reactor by manually opening Reactor Trip Breakers 1A and 1B.

08:41:46 1A SG water decreased to 16.7% resulting in a Lo-Lo SG level reactor trip signal and automatic start of the 1A and 1B MDCA Pumps.

Plant operators responded to the event adequately using plant procedures and there were no malfunctions of equipment needed to respond to the event.

This event was initially reported on March 4, 2002 as an actuation of the Unit 1 Reactor Protection and Auxiliary Feedwater Systems in accordance with the requirements of 10 CFR 50.72 (b)(2)(iv)(B) and 10 CFR 50.72 (b)(3)(iv)(A). This LER is being submitted as per the requirements of 10 CFR 50.73 (a)(2)(iv)(A) for the same system actuations described in the initial March 4, 2002 report.

INVESTIGATION

Following the event, inspections identified that the primary power supply to Cabinet 5 from Panelboard KXB was available and, as expected, the backup power supply to Cabinet 5 from Panelboard KRB was de-energized. This backup power had been secured in preparation for de-energizing Distribution Center MKB for maintenance.

Further investigation determined that PCS Cabinet 5, Card Frame 3, was de-energized. Inspection of the primary power fuse indicator for Card Frame 3 revealed that it was extinguished. The primary power fuse indicators for all other PCS Cabinet 5 card frames were illuminated. This signified that the primary power supply fuse for Card Frame 3 had blown, which was confirmed later by visual observation of the fuse. Additional inspection identified that the backup power fuse indicators for all frames within PCS Cabinet 5 were extinguished. This was expected since, as discussed previously, the backup power to PCS Cabinet 5 had been secured.

Given the blown primary power supply fuse for PCS Cabinet 5, Card Frame 3, the individual control cards on that frame were removed and tested. All cards on Frame 3 tested satisfactorily except for the following:

- Card C5-332, which contains control circuitry associated with the Unit 1 PZR PORV discharge temperature loop. The fuse for this card was found blown.
- Card C5-345, which contains control circuitry associated with feedwater level control components. Capacitor C-105 on this card was found blown due to an electrical short. The card fuse was not blown.

All blown fuses and Cards C5-332 and C5-345 were replaced. Upon energizing Distribution Center MKB and Panelboard KRB, the primary and backup power supplies were available for Unit 1 PCS Cabinet 5. The PCS circuits were then functionally verified and determined to be operating correctly.

CAUSAL FACTORS

Capacitor C-105 on Card C5-345 provides noise filtering for DC Power supplied to PCS Cabinet 5. This capacitor is installed upstream of the fuse for Card C5-345. Consequently, the electrical short on capacitor C-105 caused the upstream primary power supply fuse for PCS Cabinet 5, Card Frame 3 to blow without affecting the downstream fuse for Card C5-345. The blown primary power supply fuse for Card Frame 3 and the previous securing of its backup power supply resulted in loss of electrical power to that card frame. Since the Normal and Alternate control circuitry for both Auto and Manual operation of 1CF-32AB and the circuitry for Auto and Manual operation of 1CF-104AB is contained on Card Frame 3, these valves failed closed. With no CF supply, 1A SG water level began to decrease, resulting in the subsequent manual trip of the Unit 1 reactor and automatic actuation of the 1A and 1B MDCA Pumps.

The receipt of the Unit 1 PZR PORV high discharge temperature alarm during the event was caused by the blown fuse for Card C5-332. A direct cause for the switching of the PZR Pressure Master Controller from automatic to manual control was not identified.

Based upon the above, the root cause of this event was determined to be an electrical short on Capacitor C-105 located on Card C5-345 within PCS Cabinet 5, Card Frame 3. The direct cause of this short could not be identified. The electrical short on Capacitor C-105 resulted in a blown primary power supply fuse for Card Frame 3 and a subsequent loss of electrical power to the control circuitry for the 1A Steam Generator main feedwater supply valves, 1CA-32AB and 1CA-104AB. With no control power, these valves failed closed isolating feedwater flow to the 1A SG.

A contributing cause to this event is the PCS card design which located the C-105 capacitor upstream of the fuse for Card C5-345. This design allowed a single component failure to de-energize the entire PCS Cabinet 5, Card Frame 3.

CORRECTIVE ACTIONS

Immediate: Cards C5-332 and C5-345 were replaced.

The fuse for Card C5-332 and the primary power supply fuse for PCS Cabinet 5, Card Frame 3 were replaced.

Distribution Center MKB and Panelboard KRB were re-energized, restoring the backup power supply to Unit 1 PCS Cabinet 5. The PCS circuits were then functionally verified and determined to be operating correctly.

Planned: McGuire will evaluate possible design changes to prevent a capacitor failure from causing a reactor trip due to a loss of electrical power to process control cabinet circuitry.

SAFETY ANALYSIS

Based on this analysis, this event is not considered to be significant. At no time was the safety or health of the public or plant personnel affected as a result of the event.

Reactor trips and turbine trips are analyzed in Chapter 15 of the McGuire Nuclear Station Final Safety Analysis Report. Those analyses demonstrate that, given the plant conditions and sequence of events associated with the March 4, 2002 event, the plant design and response was adequate. Therefore, this event presented no hazard to the integrity of the Reactor Coolant System or the reactor fuel/cladding.

During the event, the unit experienced a manual reactor trip and actuation of the CA system with no complications. Feedwater flow to the SGs was maintained by the CA System, ensuring adequate decay heat removal. Given this and the availability of other plant equipment needed for initiating and maintaining adequate decay heat removal, the Conditional Core Damage Probability (CCDP) of this event is considered insignificant (estimated on the order of 6.0E-07).

The major contributors to Large Early Release Frequency (LERF), according to the McGuire PRA, are the containment bypass sequences. The manual reactor trip event does not produce sequences that contribute significantly to the containment bypass plant damage state. Therefore, the impact on LERF is very small.

Given the above, this event is considered to be of no significance with respect to the health and safety of the public.

ADDITIONAL INFORMATION

The event described in this LER is not considered to be the result of a human performance problem. A review of McGuire events for the past three years did not identify any event that had a similar sequence of events or causes.